ACCESSION #: 9607110221

LICENSEE EVENT REPORT (LER)

FACILITY NAME: SURRY POWER STATION, Unit 2 PAGE: 1 OF 5

DOCKET NUMBER: 05000281

TITLE: Turbine/Reactor Trip due to High Level in the Steam

Generator

EVENT DATE: 06/06/96 LER #: 96-004-0 REPORT DATE: 07/02/96

OTHER FACILITIES INVOLVED: Surry Unit 1 DOCKET NO: 05000280

OPERATING MODE: POWER LEVEL: 16%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: D. A. Christian, Station Manager TELEPHONE: (804) 357-3184

COMPONENT FAILURE DESCRIPTION:

CAUSE: SYSTEM: COMPONENT: MANUFACTURER:

REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On June 6, 1996, power escalation was in progress for Unit 2 upon completion of a scheduled refueling outage. Control Room Operators were transferring from manual feedwater flow control with the Main Feedwater Regulation Valve (MFRV) Bypass Valves to automatic feedwater control with the MFRVs. At 2344 hours, with Unit 2 at 16% power, a high-high Steam Generator B level signal automatically tripped the running main feedwater pump and the main turbine. The turbine trip generated a reactor trip signal. The Reactor Protection System actuated and functioned as designed, and all control rods inserted into the core. Following the trip, Control Room Operators acted promptly to place the plant in a safe, hot shutdown condition in accordance with the

proper procedures. The Shift Technical Advisor calculated the shutdown margin and monitored the critical safety function status trees to verify that the unit conditions were acceptable. Plant response was as expected and the unit stabilized at hot shutdown.

During this event the health and safety of the public were not affected. This event is being reported pursuant to 10 CFR 50.73(a)(2)(iv).

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1.0 DESCRIPTION OF THE EVENT

On June 6, 1996, power escalation was in progress for Unit 2 upon completion of a scheduled refueling outage. Reactor power was 16% based on Reactor Coolant System (RCS) [EIIS-AB] Loop delta T indications, and RCS average temperature was being controlled at 553 degrees Fahrenheit using control rods in manual and Main Steam Dump Valves [EIIS-SB-TCV] in the steam pressure mode. The main Turbine [EIIS-TA] was rotating at 1800 rpm with turbine control in the operator automatic mode. Preparations were being made to close the Main Generator [EIIS-TB] output breakers.

Between 2331 and 2333 hours, Control Room Operators were transferring from manual feedwater flow control with the Main Feedwater Regulation Valve (MFRV) [EIIS-SJ-ISV] Bypass Valves to automatic feedwater control with the MFRVs. MFRVs A and C were opened first. Steam Generator (SG) [EIIS-AB-SG] water levels were oscillating in all three SGs. The Reactor Operator (RO) assigned to SGs A and B was stabilizing the level on SG A when he noticed SG B level increasing. MFRV B was verified closed, and the demand for the MFRV Bypass Valve B was lowered to reduce feedwater flow. Level continued to increase, so demand was lowered to fully close the MFRV B Bypass Valve. Level continued to increase.

At 2344 hours, a high-high SG B level signal automatically tripped the running Main Feedwater Pump [EIIS-SJ-P] and the Main Turbine. The turbine trip generated a reactor trip signal. The Reactor Protection System (RPS) actuated and functioned as designed, and all control rods inserted into the core. The Motor-Driven Auxiliary Feedwater Pumps [EIIS-BA-P] automatically started as designed. Control Room Operators acted promptly to place the plant in a safe, hot shutdown condition in accordance with the proper procedures. The Shift Technical Advisor calculated the shutdown margin and monitored the critical safety function status trees to verify that the unit conditions were acceptable. Individual Rod Position Indicator (IRPI) [EIIS-AA-ZI] F-8 initially indicated 13 steps then gradually drifted down to less than 10 steps.

The RCS cooled down below the 547 degree Fahrenheit (no load temperature) and reached a minimum of 539 degrees Fahrenheit. RCS temperature subsequently stabilized at 547 degrees Fahrenheit after the Main Steam Dump Valves closed and the Auxiliary Feedwater (AFW) Pumps [EIIS-BA-P] were secured.

In accordance with 10 CFR 50.72(b)(2)(ii), a 4-hour non-emergency report to the Nuclear Regulatory Commission was made at 0325 hours due to the automatic RPS actuation. This event is being reported pursuant to 10 CFR 50.73(a)(2)(iv) due to automatic actuation of the RPS.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

Upon receipt of the reactor trip signal, the RPS actuated and functioned as designed, and all control rods inserted into the core. Control Room Operators acted promptly to place the plant in a safe, hot shutdown condition in accordance with the proper procedures. The Shift Technical Advisor calculated the shutdown margin and monitored the critical safety function status trees to verify that the unit conditions were acceptable. Plant response was as expected and the unit stabilized at hot shutdown. No conditions adverse to safety resulted from this event and the health and safety of the public were not affected.

3.0 CAUSE OF THE EVENT

The cause of the reactor trip was a high level in SG B. This level swell occurred when the SG A stopped steaming and SG B picked up steam flow. This level swell combined with the initial level in the SG B caused the level to increase above the 75% high-high SG level trip setpoint. In response to increasing level in SG B, the RO first lowered and then attempted to secure feedwater flow 30 seconds before the trip.

Investigations following the subsequent startup revealed that erratic Steam Dump Valve operation also contributed to the trip. Several seconds of oscillating steam dump demand occurred at approximately 9 minutes and again at 3.5 minutes before the trip. Steam Dump Valves 2-MS-TCV-205A and B oscillated in response to the demand signal. These steam dump oscillations increased the magnitude of the SG oscillations. Also, a zero shift in the controller for 2-FW-FCR-255B MFRV Bypass Valve may have contributed to the level increase in SG B.

The design of the Feedwater Control System requires manual operation of the MFRV Bypass Valve during startup. There are numerous variables involved in controlling SG levels; these variables along with the equipment malfunctions discussed in the previous paragraphs, resulted in the Control Room Operators being challenged to the point where SG level could not be successfully controlled.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Following the trip, Control Room Operators acted promptly to place the plant in a safe, hot shutdown condition in accordance with the proper procedures. The Shift Technical Advisor calculated the shutdown margin and monitored the critical safety function status trees to verify that the unit conditions were acceptable. Plant response was as expected and the unit stabilized at hot shutdown.

The SGs were initially fed by the AFW System and were later transferred to the MFRV Bypass Valves. Decay heat was removed by the Main Steam Dump Valves discharging to the Main Condenser.

5.0 ADDITIONAL CORRECTIVE ACTION(S)

Investigations following the subsequent startup revealed that Steam Dump Valve operation was erratic while in the steam pressure mode. This erratic operation occurred prior to the trip in the form of spikes in the demand to 2-MS-TCV-205A and B. During power ascension on June 7, 1996, the steam dump valves were observed to modulate in response to RCS temperature changes. Subsequent investigation by I&C determined that both RCS average temperature and steam pressure voltages were being fed to the valves due to a faulty relay in the steam dump control system. This relay was replaced. The steam dump valves were tested in both the RCS average temperature and steam pressure modes with satisfactory results.

During startup, MFRV Bypass Valve 2-FW-FCV-255B was noted as requiring less demand than the MFRV Bypass Valve A to provide the same flow (e. g., 60% demand on MFRV B provided the same flow as 80% demand on MFRV A). The demand was lowered to zero for MFRV B Bypass Valve in order to isolate feedwater flow to the SG B but level continued to increase. Following the trip, I&C technicians determined that the valve was not going fully closed on demand; however, when instrument air to the positioner was isolated, the valve would move further in the closed direction. Further investigation determined a slightly elevated pneumatic output from the Electric/Pneumatic (E/P) transducer. When the cover was removed from the E/P transducer, the input signal wires were found lying against the force beam causing elevated pneumatic output. The wires were relocated away from the force beam and the valve fully closed with E/P input at its minimum.

Further investigation of the MFRV Bypass Valve leakage and the Steam Dump Valve erratic operation is ongoing. Further corrective action related to these items may be recommended as a result of this investigation and will be tracked by the corrective action process.

A calibration check was performed on IRPI F-8 and it was found to be indicating high. IRPI F-8 was adjusted satisfactorily.

6.0 ACTIONS TO PREVENT RECURRENCE

None.

7.0 SIMILAR EVENTS

LER S2-86-003 - Turbine trip/reactor trip from high SG Level due to MFRV bypass valve failing to close on demand

8.0 ADDITIONAL INFORMATION

Unit 1 was operating at 100% during this event.

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10CFR50.73

Virginia Electric and Power Company

Surry Power Station

5570 Hog Island Road

Surry, Virginia 23883

July 2, 1996

U. S. Nuclear Regulatory Commission Serial No.: 96-347

Document Control Desk SPS: JDK/VLA

Washington, D. C. 20555 Docket No.: 50-281

License No.: DPR-37

Dear Sirs:

Pursuant to Surry Power Station Technical Specifications, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to Surry Power Station Unit 2.

REPORT NUMBER

50-281/96-004-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Management Safety Review Committee for its review.

Very truly yours,

D. A. Christian

Station Manager

Enclosure

pc: Regional Administrator

101 Marietta Street, NW, Suite 2900

Atlanta, Georgia 30323

M. W. Branch

NRC Senior Resident Inspector

Surry Power Station

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